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EXPERIMENTAL NEUTRON FLUX DISTRIBUTIONS IN A GRAPHITE ASSEMBLY WITH AN INTERNAL CAVITY

By Thomas A. Fox, Donald F. Shook, Robert A. Mueller,
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Lewis Research Center
Cleveland, Ohio

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SUMMARY

Experimental neutron flux distributions have been measured in a graphite sigma pile that was modified to form an internal cavity region. A 12-inch cubic cavity was arranged near the geometric center of a graphite cubic assembly approximately 52 inches on a side. A polonium-beryllium neutron source was located at the center of the cavity, and bare and cadmium-covered indium foils were used to determine the neutron flux distributions in both the cavity and moderator regions.

Experimental flux distributions are presented for several cavity cases as well as some solid-graphite assembly cases. The cavity cases include a boron trifluoride gas filled cavity configuration to simulate a uranium hexafluoride gas filled cavity and an air-filled cavity configuration to simulate a void. The solid-graphite configuration was used for a reference case.

An analytical study was made of the boron trifluoride gas cavity case using a spherical-geometry model modified to approximate the actual cubic geometry of the experiment. Good agreement was obtained between the experiment and calculations for thermal energy neutrons, and fair agreement was obtained for indium resonance energy neutrons.

INTRODUCTION

Considerable interest has existed for several years in cavity-type reactors. The term "cavity reactor" is used herein in its most general meaning to include any reactor having an internal region of very low density surrounded by an external reflector moderating region. The basic concept of a cavity reactor that is proposed has a fertile region of low density composed of possibly uranium hexafluoride gas surrounded by a good neutron moderating medium such as beryllium, heavy water, or graphite for an external reflector (refs. 1 and 2). Another type of cavity reactor has been proposed for a research reactor with a large irradiation facility. This cylindrical reactor has the fuel in solid form located in an annular ring between a large internal void test region and an external moderating region (refs. 2 and 3). Such a reactor might be classified more precisely as an externally moderated reactor with an internal void region. Applications of cavity-type reactors for high-performance propulsion systems have also been proposed. Such proposals include a coaxial flow reactor for rocket applications (ref. 4) and a plasma core reactor for possible space applications (ref. 5). Despite the variation in details and the proposed uses of these reactors, they have in common an interior region of low-density material surrounded by an exterior region composed of a good moderating material.

The analytical studies performed to date use idealized geometries and simplified models for neutron diffusion and slowing down. There have been few experiments performed to check such calculations. Of particular interest is the spatial distribution of neutrons in systems containing a low-density absorptive region adjacent to a moderator region such as graphite. Although the cavity is usually quite large physically, it can be small in terms of the nuclear parameters of the cavity region such as slowing-down length and neutron total mean free path.

The work reported herein was undertaken to provide experimental data regarding neutron diffusion in cavity media. An insight into the diffusion properties of cavity reactors may be gained through the use of a polonium-beryllium neutron source placed in a cavity within a graphite assembly. The primary objective of this work was the measurement of neutron flux distributions with the cavity filled with a neutron absorbing gas. The boron trifluoride gas was especially good for this purpose because it closely approximates the thermal neutron absorption properties of uranium hexafluoride gas. The actual experiment consisted of a 12-inch cubic cavity centrally located in a cube of graphite approximately 52 inches on a side. A limited analytical investigation of the experiment with an absorbing gas in the cavity was made, and the results are compared with the experimental data.

DESCRIPTION OF EQUIPMENT AND EXPERIMENTAL PROCEDURE

Although the primary objective of this work was the measurement of neutron flux distributions with the BF_3-filled cavity in the graphite assembly, various other measurements were required as well. Measurements were made for the following experimental cases:

(1) Solid-graphite parallelepiped, approximately 52 by 52 by 56 inches

(2) Graphite assembly with 12-inch cubic void in center

(a) No liner, air-filled cavity

(b) Aluminum liner, air-filled cavity

(c) Aluminum liner, BF_3-gas-filled cavity

The measurements consisted primarily of bare and cadmium-covered indium-foil activations by neutrons from a Po-Be source. Additional work was done with indium foils covered with both cadmium and indium. The cadmium-and-indium-covered indium data were taken to determine if the activity of the cadmium-covered indium foils due to neutrons significantly above the 1.4-ev indium resonance affected the measured distribution.

Neutron Source and Detecting Foils

The Po-Be neutron source used for the graphite pile measurements and the foil calibrations was a standard model Mound Laboratory capsule in the shape of a cylinder. The source material was doubly canned in cold-rolled steel and stainless-steel containers and had external dimensions of 0.7 inch in length and 0.7 inch in diameter. The original activity of the source was 10 curies but had decayed through more than two half-lives prior to these tests. During this work the source strength varied from about 2 curies to less than 1 curie or from about 4×10^6 to about 1×10^6 neutrons per second.

The indium foils used for most of this work were of chemically pure indium with dimensions of 1 centimeter by 1 centimeter by 95 milligrams per square centimeter thick. This corresponds to a thickness of approximately 0.005 inch. Unalloyed indium is quite soft and required protection in some applications during this work. Protection was accomplished as needed by packaging in aluminum foil, scotch tape, or tissue paper. Such protective packaging did not produce any measurable effect on the data being taken. The foils had been calibrated with respect to one another prior to the start of this work by a standard technique and a set of closely identical foils chosen for this study.

The cadmium covers used in this work had a nominal thickness of 0.020 inch and were constructed with overlapping edges to prevent edge leakage or neutron streaming in any fashion into the enclosed volume. The indium foils used in the cadmium-plus-indium-covered indium foils were the same as previously described. A total of five indium foils were stacked together, placed inside a cadmium cover, and exposed; and the activity of the middle one was counted. The cadmium-plus-indium-covered indium therefore had 0.020 inch of cadmium and 0.010 inch of indium on each side.

Whenever possible, the foils were activated to saturated values by exposing them overnight. In some cases, notably the cases with the cavity filled with BF_3 gas, it was advantageous to make shorter exposures leading generally to about 80 to 90 percent of the saturated activity. In all cases, the foil activities were counted in gas-flow proportional counters having a 2π geometry.

Graphite Assembly

The graphite assembly used in these measurements was made up of AGOT reactor grade graphite stringers each having dimensions of 4 by 4 by 51 inches and a mean density of 1.69 grams per cubic centimeter. These stringers were stacked with alternate layers running normal to each other, which made the edge of the pile irregular with an actual average horizontal side dimension of $51\frac{1}{2}$ inches. A total of 14 layers were stacked to have a vertical height of 56 inches. A good picture of the graphite assembly can be obtained by referring to figure 1. This shows the graphite assembly modified to give a central cavity region. In the unmodified graphite stack no stringers protruded from the outside face.

Several of the stringers in this assembly were modified to accommodate a neutron source and/or foils for activation. Because of the crossed manner in which the stringers were stacked, virtually any graphite stringer (regular or special) could be pulled out of the assembly like a drawer. Since the various pieces were interchangeable, very good flexibility could be achieved in the positioning of the source and the foils.

The flux distribution measurements were made with the source located slightly above the geometric center of the assembly. The source was located in the center of the seventh layer from the top of the assembly. This made the source 26 inches from the top, 30 inches from the bottom, and approximately 26 inches from the other edges. Since the vertical flux traverses were taken only above the source, the horizontal and vertical traverses were over essentially the same size region of graphite, and the stack can be treated like a 52-inch cube of graphite. The source was located near the center of the stack by cutting one of the graphite stringers into two equal pieces (4 by 4 by $25\frac{1}{2}$ in.), drilling a hole for the source in the end of one half piece, and inserting the half stringer with source in the appropriate position in the assembly. This hole was drilled so that the source was located with its axis on a horizontal plane as shown in figure 2.

The other half stringer was also modified to position foils precisely in a manner that permitted their use in making horizontal flux distribution measurements. This half stringer was placed opposite the half stringer containing the source, the two special half-size stringers replacing one full-size stringer. Figure 2 illustrates the construction and relative arrangement of the half stringers when in use. The source and foils were not in exactly the same plane when used in this manner, since the foils were in slots near the top surface of the one special stringer and the source was centered in the end of the other special stringer.

In addition to the special stringers just described, several other graphite pieces have been modified to accommodate foils that lay flat in the horizontal plane (see fig. 1). These were used to make flux distribution measurements along a vertical axis. Different vertical positions were achieved by adjusting the location of these graphite pieces.

Graphite Assembly with 12-Inch Cubic Cavity

A 12-inch cubic cavity was made in the center of the graphite assembly by cutting additional stringers in half like the special stringers described and arranging them as shown in figure 1. The stringers were allowed to protrude out of the opposite faces rather than be cut off flush so that the assembly could be restored to essentially its original solid condition quickly and easily.

The source was located in the center of the cavity by means of a 3/16-inch thin-wall aluminum tube with a cup on the end. This rod was fastened to the center of the end face of one of the half-sized stringers as shown in figure 1. The source was oriented with the cylinder axis in a horizontal plane as in the solid-graphite case.

The special graphite piece containing the vertical slots to handle foils was placed opposite the source. A wire of 2S aluminum 1/16 inch in diameter was installed in this piece so that it extended into the cavity as shown in figure 1. This wire was notched at 1-inch intervals so that foils could be hung in the cavity region in line with foils placed in the machined slots. The wire was 8 inches long, permitting measurements from the graphite wall to beyond the center of the cavity if necessary. Aluminum foil and scotch tape were used at different times for foil hangers. Foils were also activated in the special graphite piece as well as in the cavity during these measurements. The same technique was used for irradiating foils in the special stringer as was described in the previous section on the solid-graphite stack.

Aluminum Liner for 12-Inch Cubic Cavity

In order to fill the cavity region with BF_3 gas to provide a low-density neutron absorbing medium, it was necessary to construct a reliable container for several reasons: BF_3 gas is very toxic, is quite corrosive if any moisture is also present, and would contaminate the graphite seriously if it leaked. Since the container had to have no objectionable nuclear properties as well as contain the gas, it was constructed of 24S aluminum.

The aluminum liner was built with 1/16-inch-thick walls and sealed with welds at the edges (see fig. 3). One face had a 6-inch screwcap sealed with a gasket and designed to fit flush externally. This cap supported a 3/4-inch hollow source tube that permitted the location of the neutron source in the center of the box without the source capsule coming in contact with the BF_3 gas. The aluminum wire used to position the indium foils was installed on the inside face of the box opposite the screwcap. Both the source location and the foil locations duplicated as nearly as possible the positions used in the previous cavity case. The box was flushed with dried nitrogen gas to assure dryness of the interior before filling with BF_3. It was found that the BF_3 could be contained satisfactorily as long as no moisture was present. The box was filled outside the graphite pile and checked for leaks before insertion. Small tubes from the box were brought out of the pile to a pressure gage and manometer to allow continuous monitoring of the pressure of the gas in the box during runs. No gas was introduced or removed while the box was in the cavity. All heavy structural perturbations such as the screwcap, source tube, and gas tubes were kept on one side of the box. The foil activation measurements were made on the opposite side of the source from which the perturbing structure was located.

It was necessary to package in scotch tape the foils that were located in the BF_3 atmosphere. The tape also served as the hangers for the foils. Since the foils had to be inserted in the box prior to filling with gas and insertion of the box in the graphite, considerable care in handling the box was necessary to keep from dislocating the foils. The advantage, if not necessity, of being able to leave the source out of box until the assembly was completed is quite apparent.

In general, the foils were not exposed overnight or to saturation when using BF_3 in the cavity, although most runs resulted in exposures to 80 percent or better of saturated values. All data were obtained from one bottle of BF_3 gas in which the boron was present in naturally occurring isotopic percentages. The box was always filled to a few millimeters greater than atmospheric pressure. There was a tendency to lose gas pressure with time, although numerous leak checks and repeat tests of earlier experiments did not reveal any leakage from the box.

One possible explanation for this effect is a slow chemical reaction between the BF_3 gas and the inside surface of the liner. The variation in initial BF_3 pressure was less than 2 percent from one run to the next, and the variation during a run was less than 5 percent. No corrections due to the possible variation in gas density were made in these tests.

EXPERIMENTAL RESULTS

Foil Activation Data

This section presents the experimental results obtained for the various cases in the following order:

(1) BF_3-gas-filled cavity

(2) Air-filled cavity

(3) Solid graphite

A comparison of the thermal neutron flux data for these cases is then presented and discussed. The section is concluded by a discussion of the data reduction and experimental precision.

BF_3-gas-filled cavity. - The data obtained from the experiments with BF_3 gas contained in the cavity include bare and cadmium-covered indium foil activation measurements to obtain the neutron flux distributions in a horizontal direction. These data are presented in figure 4 with continuous curves drawn through the respective foil activations. The thermal neutron flux distribution as obtained from the difference of these two curves has also been plotted in figure 4. The smooth curve drawn through the cadmium-covered data is based on a combination of these data and the cadmium-covered data obtained with air in the cavity as presented later. This curve assumed that BF_3- and air-filled cavities both appear to be voids with regard to higher energy neutrons.

Air-filled cavity. - The data obtained from the air-filled cavity experiments include bare and cadmium-covered indium foil activation measurements to obtain the neutron flux distributions in both the horizontal and vertical directions. These data are presented in figure 5 along with the thermal neutron flux distribution in the horizontal direction. The thermal neutron data in this case were obtained by taking the difference between the actual bare and cadmium-covered experimental data at the respective points of measurement. The differences between the horizontal and vertical flux traverses are discussed later.

Solid-graphite pile. - In order to obtain the thermal neutron flux distribution for the solid-graphite standard case, a number of measurements were necessary. The results of the measurements carried out with the basic graphite pile are presented in figures 6 and 7. These data include bare foil, cadmium-covered foil, and cadmium-plus-indium-covered foil traverses in the horizontal direction.

Figure 6 shows the bare foil, cadmium-covered foil, and thermal flux traverses in the horizontal direction. Figure 7 shows the cadmium-covered foil (from fig. 6), the cadmium-plus-indium-covered foil, and the indium resonance flux traverses (difference between cadmium-covered and cadmium-plus-indium-covered data) in the horizontal direction. The vertical traverses in the solid-graphite assembly case are very similar to the horizontal traverses and are not included.

Comparison of thermal flux distributions. - The thermal data from flux traverses made with the cavity filled with BF_3 gas, the cavity filled with air only, and the graphite pile with no cavity are compared in figure 8. The solid-graphite pile containing no cavity shows a typical gaussian-shaped curve. The experimental data have some small deviations not of importance here, but which are discussed later. The thermal flux in the air-filled cavity near the source is a little more than half the magnitude of the flux present when there is graphite in this region. The slight decrease in thermal flux in the center of the cavity cases compared with the flux at the cavity interface is attributed mostly to the absorption of thermal neutrons by the neutron source container. The thermal flux distribution in the BF_3-gas-filled cavity indicates a significant absorption of thermal neutrons with a general flux depression to about 30 percent of the thermal flux present in the air-filled cavity case. However, there seems to be no significant difference in the relative slope of the fluxes across the respective cavity cases as shown by the nearly flat thermal flux distribution within the cavity for all cavity cases. Since the microscopic thermal neutron absorption cross section σ_a of BF_3 with naturally occurring boron isotopic content reasonably approximates the microscopic thermal neutron absorption cross section of 93 percent enriched UF_6 (σ_a of BF_3 = 755 barns; σ_a of UF_6 = 645 barns), the thermal flux in a graphite reflected cavity reactor with the cavity filled with UF_6 at atmospheric pressure should be very uniform.

The crossover of the thermal neutron flux in the graphite region beyond about 16 inches from the source is readily explained (see fig. 8). This crossover occurs because of different outside boundaries in the solid-graphite case and the cavity cases. In the region where the horizontal flux traverses were measured in the cavity cases, the outside boundary is extended 6 inches in the process of forming the cavity. Hence, the flux measurements extend to an outer boundary interface considerably farther from the source in the cavity cases. A similar effect is shown for the horizontal and vertical bare-foil flux distributions taken in the air-filled cavity case (see fig. 5) although no crossover of curves takes place. The horizontal bare-foil traverse in figure 5 is represented by the solid line and the vertical traverse by a dashed line. The difference between the curves is not significant until the distance from the source gets greater than about 18 inches.

Although the extra graphite in the cavity cases increased the thermal flux present at larger distances from the source in this particular horizontal measurement, it is interesting to note that for comparable cadmium-covered cases the vertical distribution is essentially indistinguishable from the horizontal distribution. Neutrons with energies in the epicadmium range do not have their distributions significantly altered by the effect of the small addition to the graphite assembly in a local region.

Effect of aluminum cavity liner. - Figure 9 shows the comparison of the thermal flux associated with an air-filled cavity both with and without the aluminum liner. The presence of the aluminum liner caused a reduction of more than 10 percent in the thermal flux in the cavity region. Since the macroscopic neutron absorption cross section of 24S aluminum is not very different from that of BF_3 gas at atmospheric pressure, the liner is assumed to have altered the BF_3-gas-filled cavity data very little from an idealized graphite - BF_3-gas configuration. For this reason the BF_3-filled cavity case was compared with the ideal air-filled cavity case rather than the air-filled aluminum liner case. The results shown in figure 9 are primarily of interest in demonstrating the effect of a very weak absorber in the cavity region.

Other observed effects. - It was observed that indium foils located within a few centimeters of the Po-Be source during overnight exposures had a 4.5-hour activity of such value as to require a correction to the observed activity. The two or three positions nearest the source in the solid-graphite and air-filled cavity cases were the only ones requiring this correction. The BF_3-gas-filled test did not involve foil exposures of sufficient length in time for such a correction to be significant. Details of this correction are given in the appendix.

After applying the correction for the 4.5-hour indium activity, there was still a very decided irregularity in the data points near the source (see figs. 6 and 7). The depression in the bare-foil data point nearest the source (fig. 6) appears to be due to a depression in the thermal flux. This is attributed to the absorption of thermal neutrons by the neutron source container materials.

Figure 7 shows a sharp peaking in the cadmium-covered and cadmium-plus-indium-covered foil data very near the source. This is attributed to a 54-minute indium activation due to uncollided and once-collided source neutrons. In a very short distance from the source all the traverses take on the characteristic gaussian-like shape such as exhibited by the indium resonance data.

The last few data points near the outside edge of the graphite pile show a tendency to give a slight upward inflection to the distribution curve. Figure 6 indicates that the bare-foil data and the thermal flux curve extrapolate to zero well beyond the expected extrapolated boundary. This upturn in flux can be explained by backscattering of neutrons from the surroundings. The graphite pile is too small to be effectively infinite for neutron diffusion measurements, and the room housing the graphite pile is not big enough to prevent reflection of neutrons from walls, ceiling, and floor back to the pile.

Reduction of Experimental Foil Data

Many factors must be taken into consideration whenever foil activation techniques are employed to measure radiation. This is particularly true where absolute foil data are desired (ref. 6). Certain simplifications were possible in this work since only relative data were needed. The corrections associated with the counting equipment in particular were greatly reduced in number. Certain experimental techniques such as the use of nearly identical foils and exposure of foils to saturated values also were used when possible to reduce the number of corrections required.

The basic corrections used in the reduction of these foil data were:

(1) Correction for the decay of the foil activity during the time between the end of foil exposure and start of counting

(2) Correction for the decay of foil activity during foil counting (important for low-level activities that require long counting times)

(3) Correction for counting equipment background

(4) Correction for the decay in neutron source intensity

(5) Correction for the fast neutron activation of foils located very near the neutron source

(6) Correction for the difference between indium foils although the foils were chosen to get a closely identical set

In addition, the following corrections were also considered:

(1) The flux depression associated with cadmium-covered foils in the BF_3 and carbon media were compared but were found to cause a difference of less than 1 percent between media.

(2) Counting losses due to counter dead time were found to be negligible for the counting rates involved.

(3) For several BF_3 runs a correction was necessary when the foils were not exposed to saturated values.

The foil data were all normalized to the saturated activity or count rate that would be present immediately after an infinite exposure. In addition, the correction for source decay normalized the data to the date of source calibration. The normalization corrections are discussed in greater detail in the appendix.

Precision of Experimental Measurements

The data presented in this report are difficult to analyze statistically in any rigorous manner. It should be noted also that the nature of the data does not require that their accuracy be known with outstanding precision for the results to have meaning. Therefore, this section is presented in general terms with little discussion or qualification. The percent variation referred to throughout is the 95-percent confidence limit (1.645 times the standard deviation). An average variation of ±1 percent, for example, means that the 95-percent confidence limits would average ±1 percent over all points for that case.

In general, the variation in the results reflected the difficulty involved in making the measurement. The solid-graphite pile data (neglecting the last two foil positions near the edge of the pile, where counts were extremely low) show an average variation of less than ±2 percent in the bare-foil measurements and less than ±4 percent for the cadmium-covered measurements. Of this variation about 1.5 and 2.5 percent, respectively, can be attributed to counting statistics alone.

In the 12-inch void cavity case the bare-foil data show an average variation of approximately 2.5 percent and the cadmium-covered data an average variation of about 4.25 percent. Of this scatter, about the same amount of variation, 1.5 and 2.5 percent, respectively, can be attributed to counting statistics alone. The increased variation beyond this is considered to be indicative of the degree of increased difficulty in making the measurements in this case.

The BF_3-gas-filled cavity case shows a rather large scatter of data compared with the other cases. This is caused by increased difficulty of measurements and more variables being involved, as well as poorer counting statistics and fewer measurements obtained. The bare-foil data show an average variation of about ±4 percent and the cadmium-covered data a ±7.5-percent average variation. The scatter of data tends to be a little greater in the cavity region than in the graphite, but the difference is not significant to the results presented in this report. Of the variation, about 2.75 and 5 percent, respectively, can be attributed to counting statistics. Despite the loss of precision in the BF_3 data, particularly in the cadmium-covered data, the significant results are not altered or seriously affected.

ANALYTICAL RESULTS FOR BF_3-FILLED CAVITY CASE

An analysis has been made of the experimental data for the case with the BF_3-gas-filled cavity only. An 18-energy-group, one-dimensional, spherical-geometry calculation was carried out on an IBM 704 computer. A spherical-geometry model is the only practical method to use in such calculations. The spatial solution employed 200 mesh points, which were chosen to give greater detail in the region near the cavity-graphite interface. The Po-Be neutron source was considered as a point source at the center of the cavity.

The macroscopic cross sections for the graphite were obtained by the method given in reference 7, using the Po-Be spectrum. The boron cross sections were considered as $1/v$ (where v is neutron velocity) for absorption and the microscopic scattering cross section was taken to be constant at 4 barns. The Po-Be spectrum used was the spectrum given in reference 8 in which a 3-percent component below 1 Mev is assumed. The neutron age to indium resonance in an infinite graphite medium for Po-Be neutrons of this spectrum is calculated to be 433 square centimeters. The 18-group solution not only calculated the thermal neutron flux distribution, but energy group 17 roughly approximated (0.125 to 60 ev) the resonance region of indium. Therefore, a comparison of experimental cadmium-covered flux distribution has been made with the distribution given by energy group 17.

The major difficulty in the analysis concerned translating the cubic geometry of the experiment to a reasonably equivalent spherical model as needed for calculation. A spherical model was devised which gave a good agreement for the thermal neutron distribution and a fair agreement for the epithermal neutron distribution as shown in figure 10. This model included the following approximations:

(1) The cavity-graphite interface for the horizontal flux traverse was located 6 inches from the Po-Be point source. This interface

distance was kept the same by using a 12-inch-diameter sphere in the calculations to represent the 12-inch cubic cavity in the experiment. An attempt to use equivalent volume spheres did not give results that could be compared because of boundary mismatch between experiment and calculation.

(2) The outside boundary was located at the distance where the thermal flux apparently would go to zero based on the experimental distribution between 16 and 28 inches from the source. This technique was used because no analytical method was apparent for determining where such an irregular boundary should be considered as being located.

(3) The total number of neutrons absorbed in the cavity was kept equivalent by putting the same number of BF_3 atoms in the 12-inch-diameter sphere as were present in the 12-inch cubic box in the experiment. This resulted in a higher density of atoms in the spherical approximation than is present at atmospheric pressures such as in the experiment.

The calculated flux distributions are shown as the dashed lines in figure 10. The experimental data points, which have been plotted to fit the respective curves in figure 4, are not connected with any lines. The curves are normalized to unity at the point of peak thermal flux in the graphite.

Good agreement has been obtained between the analytical and experimental data. The only significant variation in the thermal neutron flux distribution is found in the cavity region. This is to be expected since the analytical model had a 60 percent higher BF_3 atom density in the cavity. The flux decrease was therefore greater in the calculations, although the flux averaged throughout the volume of the cavity was very similar. The epithermal flux distribution as calculated for energy group 17 gives the same shape curve as the experimental cadmium-covered data but shows a 10 to 15 percent larger value quantitatively. This difference may be due to the fact that energy group 17 actually extended over a greater energy range than the indium resonance absorptions cover.

SUMMARY OF RESULTS

The specific results obtained from the experimental studies of neutron flux distributions associated with a 12-inch cubic cavity located in the center of a 52-inch cubic graphite sigma pile are:

1. Removal of the graphite moderator from the cavity region (to make an air-filled cavity) results in an average reduction of the thermal flux in the cavity region to 60 percent of the amount present in the solid-graphite pile. The flux distributions across the cavity were nearly flat in all cases.

2. Insertion of BF_3 gas at atmospheric pressure into the cavity caused further reduction of the thermal flux in the cavity region to about 20 percent of the original flux level.

3. The Po-Be source spectrum with a 3-percent component below 1 Mev does give an accurate prediction of flux distributions in media around a Po-Be source.

4. A diffusion calculation analysis does give good agreement with experiment in these kinds of media. Better results should be possible when experiment and calculation are performed in the same geometry.

Lewis Research Center
National Aeronautics and Space Administration
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APPENDIX - ADDITIONAL DETAILS ON SEVERAL CORRECTIONS REQUIRED FOR DATA REDUCTION

The corrections made in the process of data reduction have been discussed in a previous section. These are fairly standard to any foil work being carried out, and further discussion of a few details is presented here primarily for the convenience of persons not engaged in foil activation work. The raw experimental foil data obtained in these studies were normalized to the saturated activity or count rate that would be present immediately after an infinite exposure. The saturated count rate N_o (counts per unit time) of a foil is given by the expression

$$N_o = \frac{n\lambda}{e^{-\lambda t_1} - e^{-\lambda t_2}} \cdot \frac{1}{1 - e^{-\lambda T}}$$

where n is the number of counts recorded in time t_1 to t_2, λ is the radioactive decay constant of the foil, t_1 is the time from the end of exposure to the start of counting, t_2 is the time from the end of exposure to the end of counting, and T is the time of foil exposure. When the exposure time T is larger than 5 or 6 half lives of the foil, the last term on the right approaches unity and can be neglected. In the case of overnight exposure of indium, exposure times often reached as much as 20 half lives and the simplified expression without the last term could be used. In much of the data taken with BF_3 gas in the cavity foil exposures were for approximately 2 hours. The last term is then important and must be used.

A daily log of all foil exposures were kept to assure the proper correction for the decay in intensity of the Po-Be neutron source. This correction was made on a daily basis only with no attempt to account for any unit of time shorter than one day. The actual value used for the correction is given by the expression $\exp(-0.0050077 \times \text{Number of days})$. This is based on a half life of polonium of 138.387 days. All data were normalized to the source strength on the date of calibration.

Foils located within a few inches of the source were found to have a 4.5-hour activity of significant value due to activation of indium-115 by the inelastic scattering of fast neutrons. Several measurements were devoted to the determination of this activity in the foils. To do this the indium foils were exposed in the usual manner for the usual time. The foil activities were then plotted against time, and the 4.5-hour component was determined by subtraction of the 54-minute activity and background from the total. The 4.5-hour activity was normalized to the end of irradiation time. Correction factors were then determined for bare and cadmium-covered activities and for solid-graphite and cavity cases. These were then applied to the data points where the corrections would make a significant change. This correction proved necessary only on foils located within about $2\frac{1}{2}$ inches of the source, which had been exposed overnight or longer. The percentage of correction was higher for cadmium-covered data by an amount dependent on the cadmium ratio.

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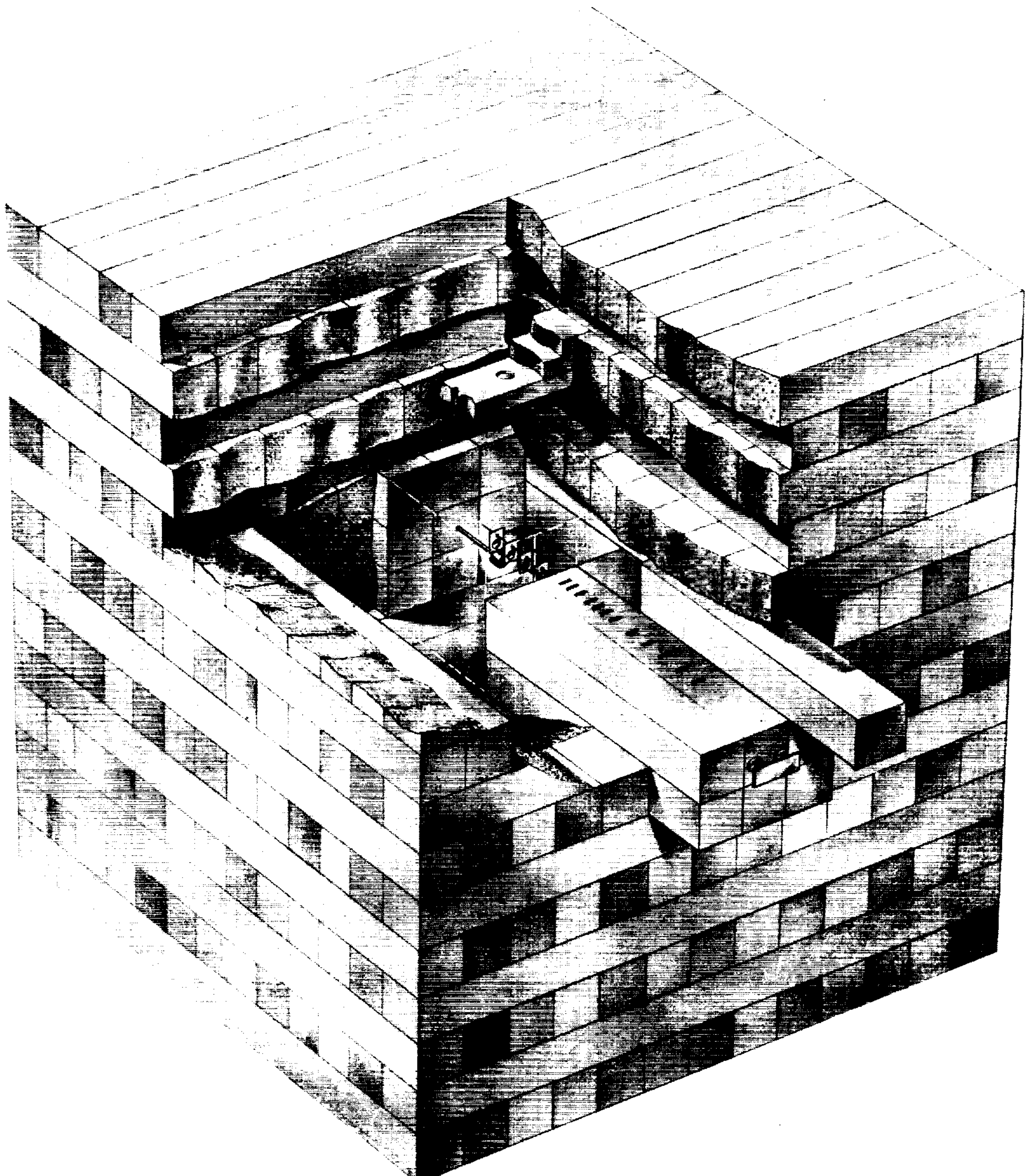

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Figure 1. - Graphite assembly modified to make central cavity region.

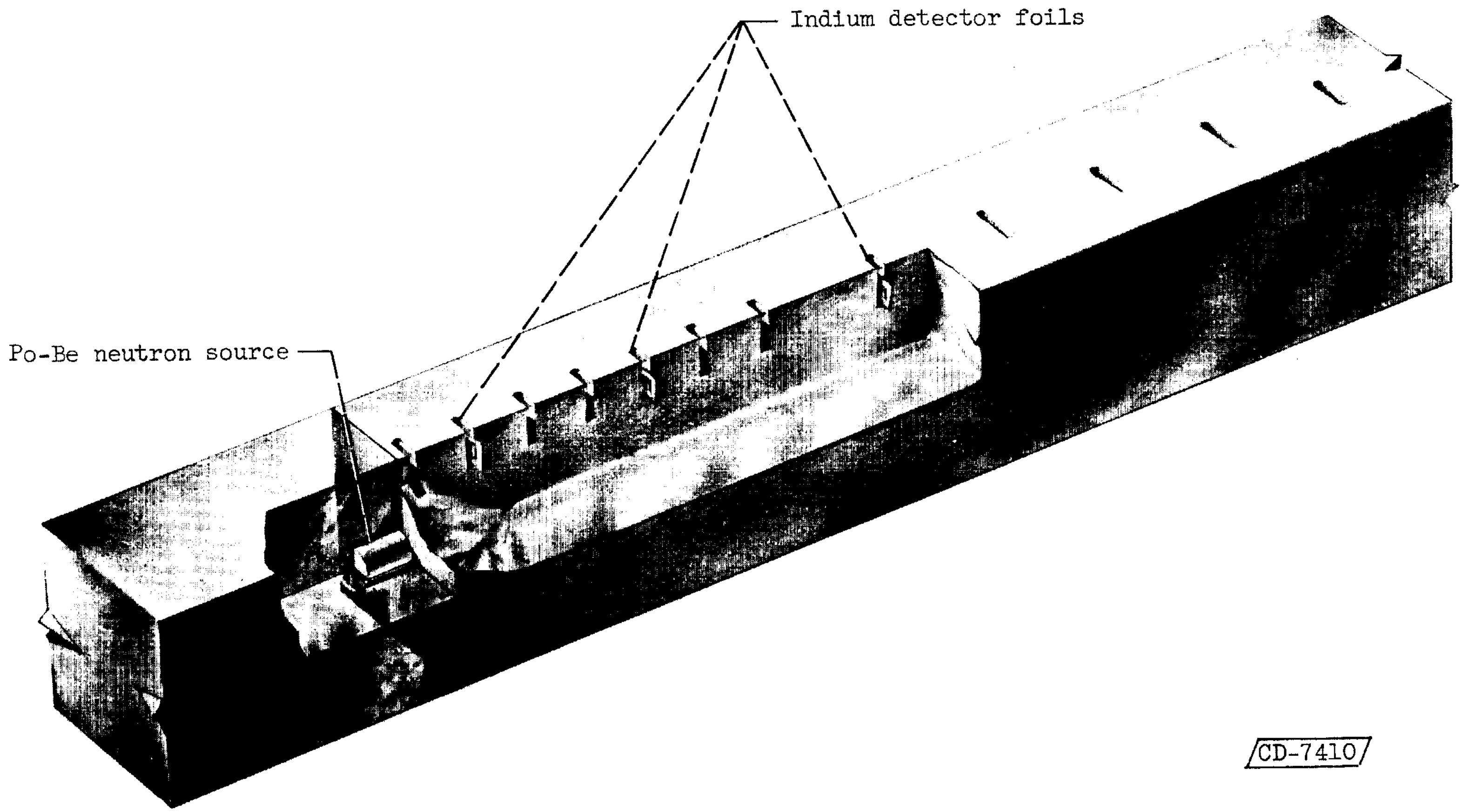

Figure 2. - Special stringers to house neutron source and indium foils for horizontal flux traverses.

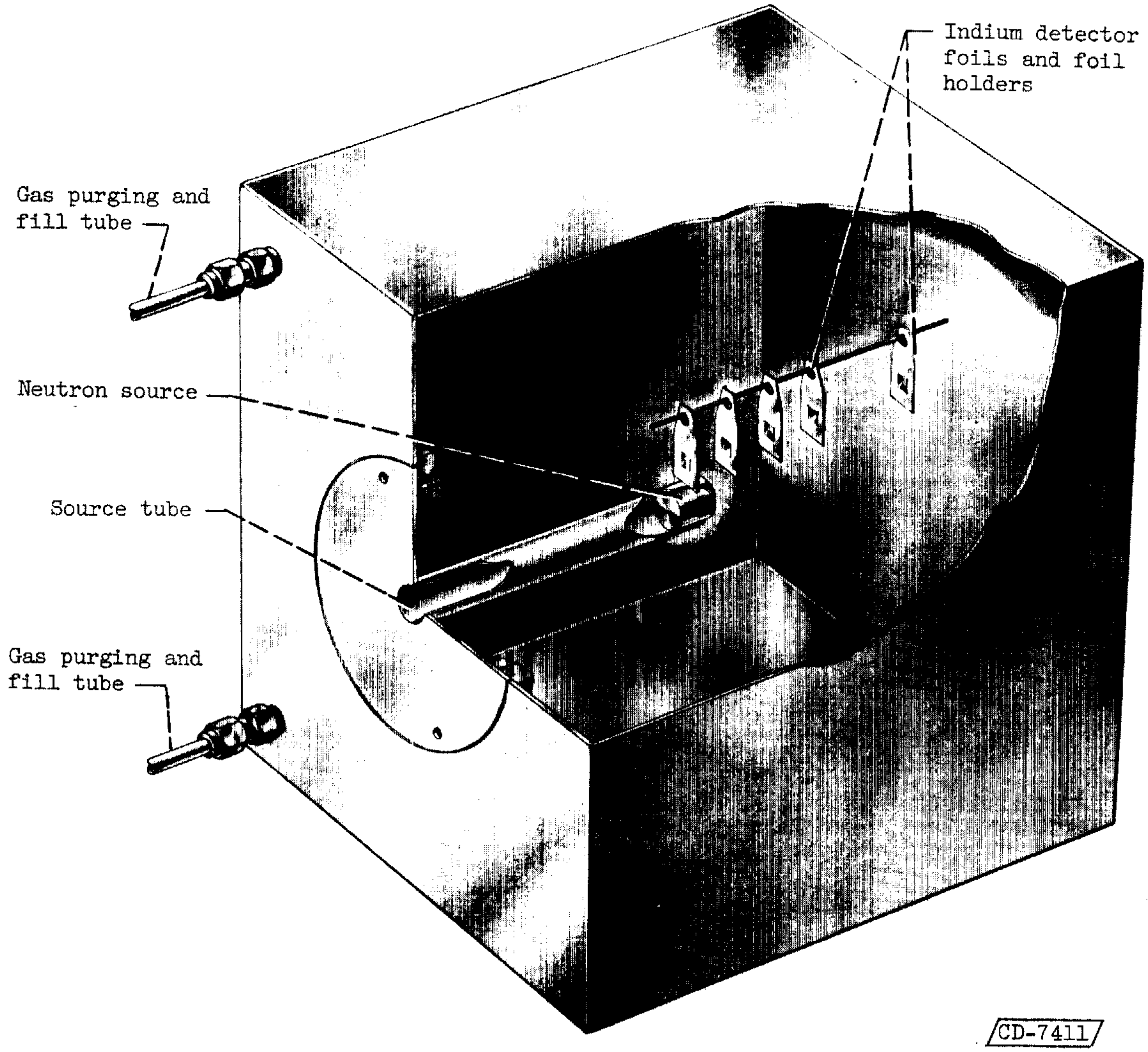

Figure 3. - 12-Inch cubic aluminum liner used to conduct tests with BF_3 in cavity.

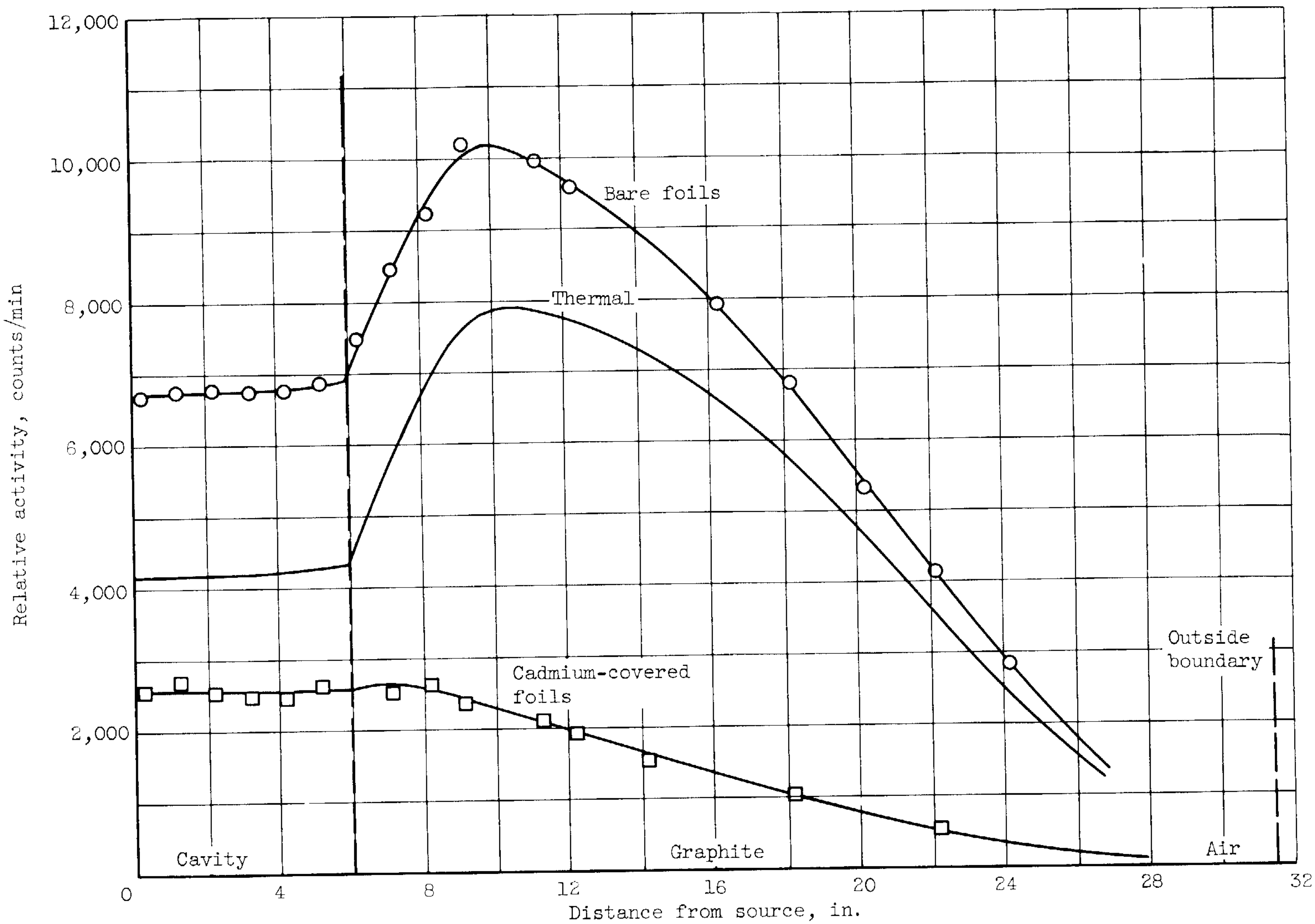

Figure 4. - Indium foil activation distributions in graphite pile with 12-inch cubic cavity filled with BF_3.

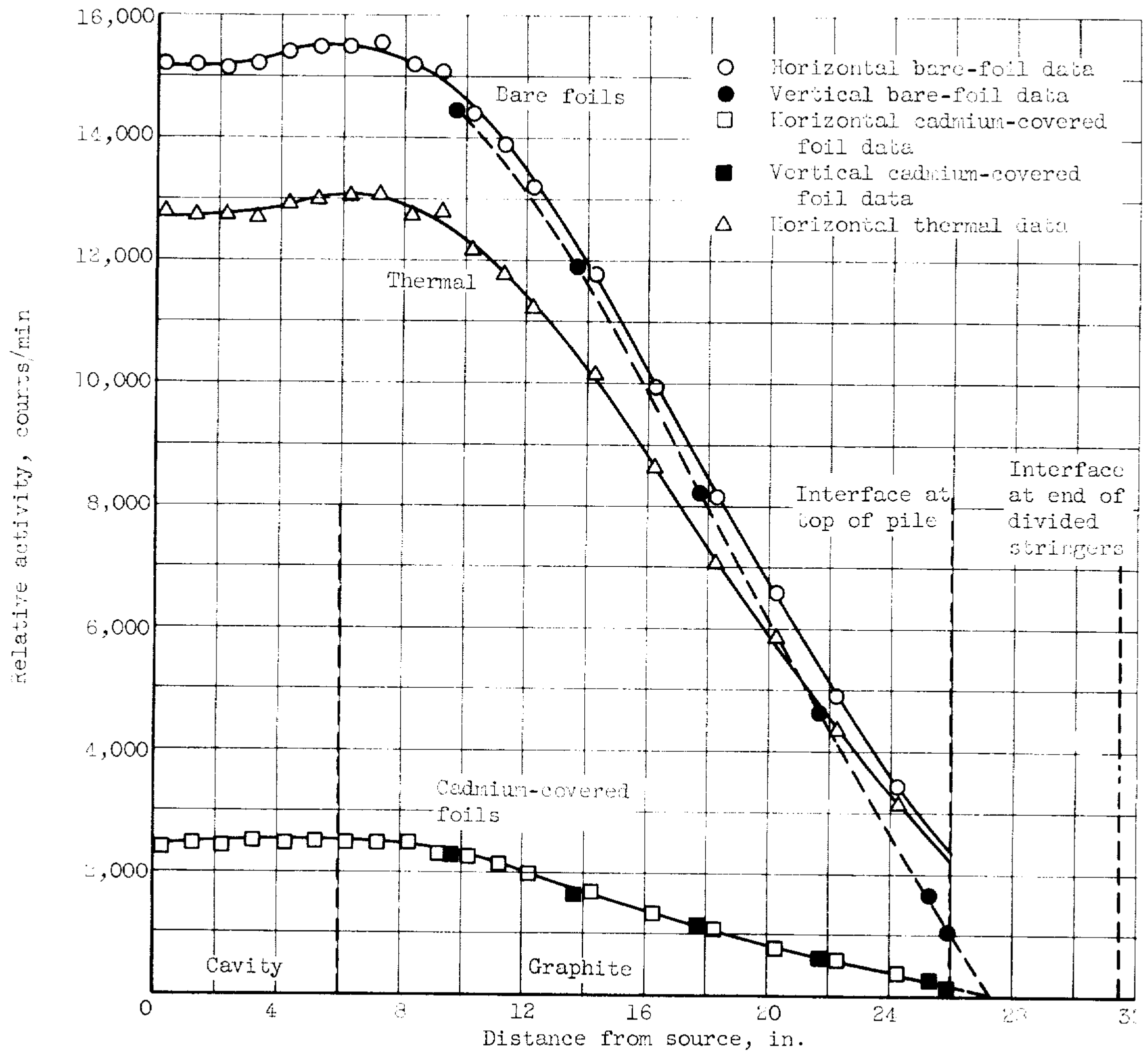

Figure 5. - Indium foil activation data for graphite pile with 12-inch cubic cavity (air-filled or void).

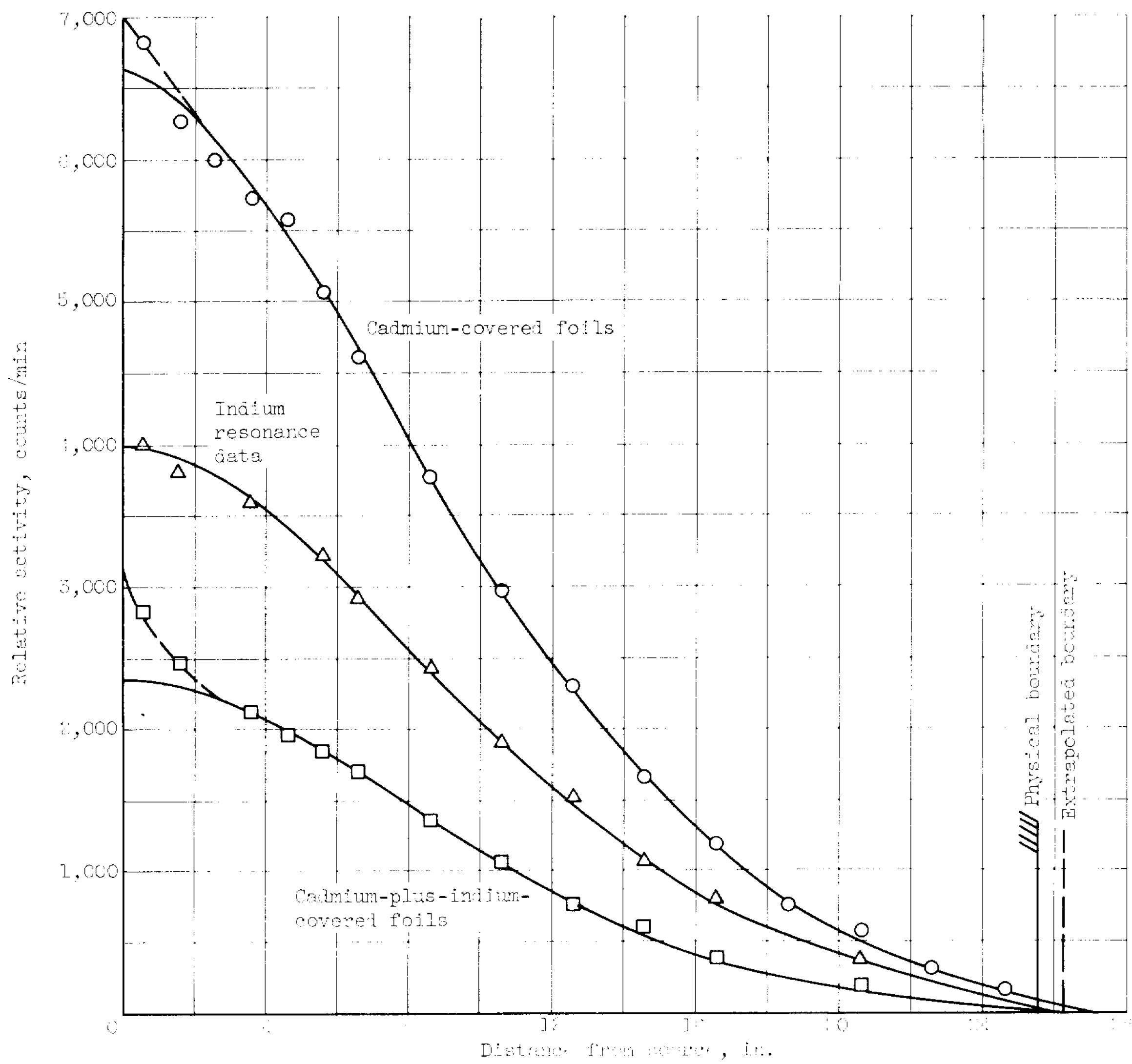

Figure 7. - Indium foil activations for cadmium-covered and cadmium-plus-indium-covered foils in solid-graphite pile ([illegible] by [illegible] by [illegible] in.).

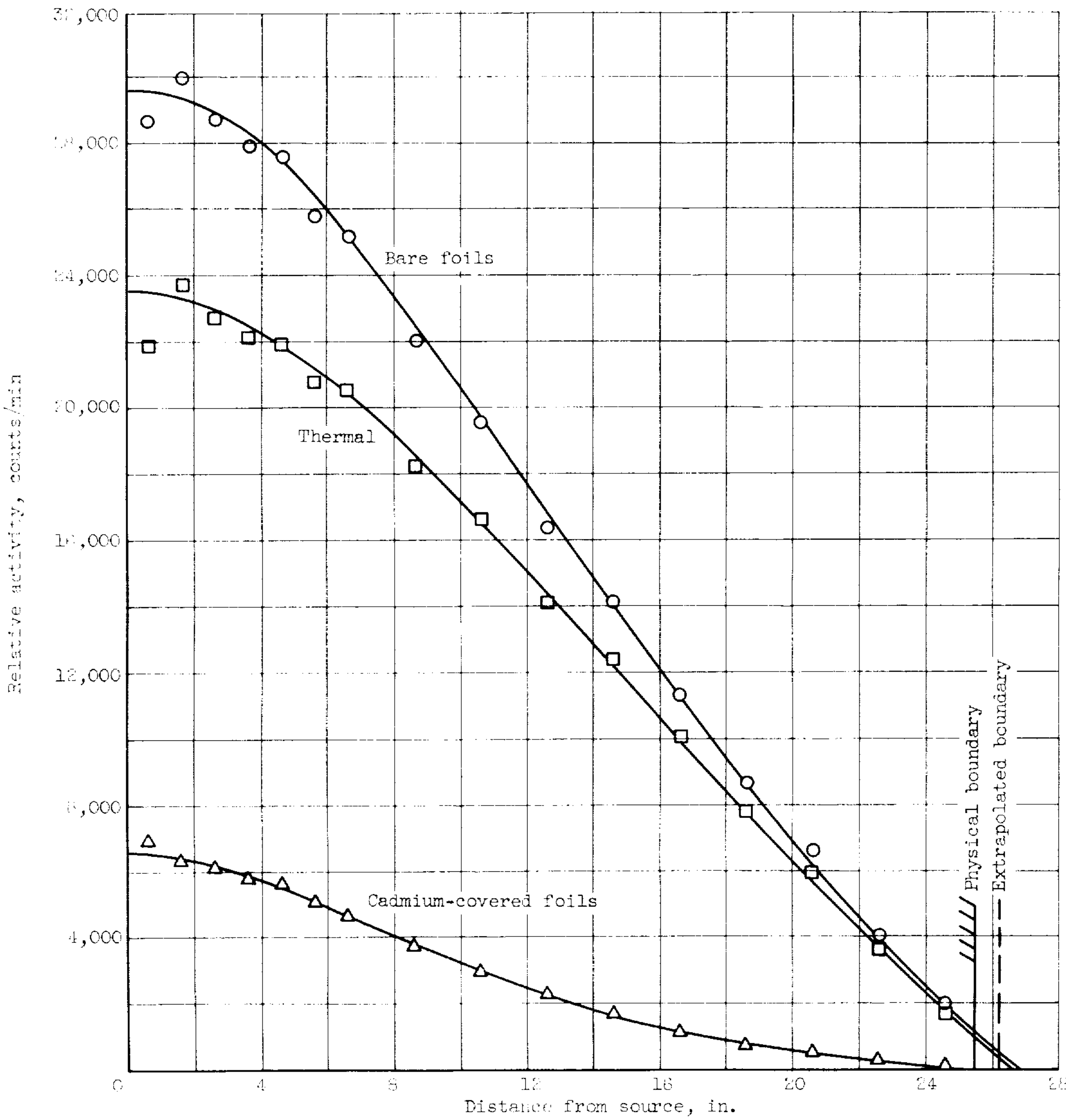

Figure 6. - Indium foil activations for bare foil and thermal neutron activations in solid-graphite pile (52 by 52 by 56 in.).

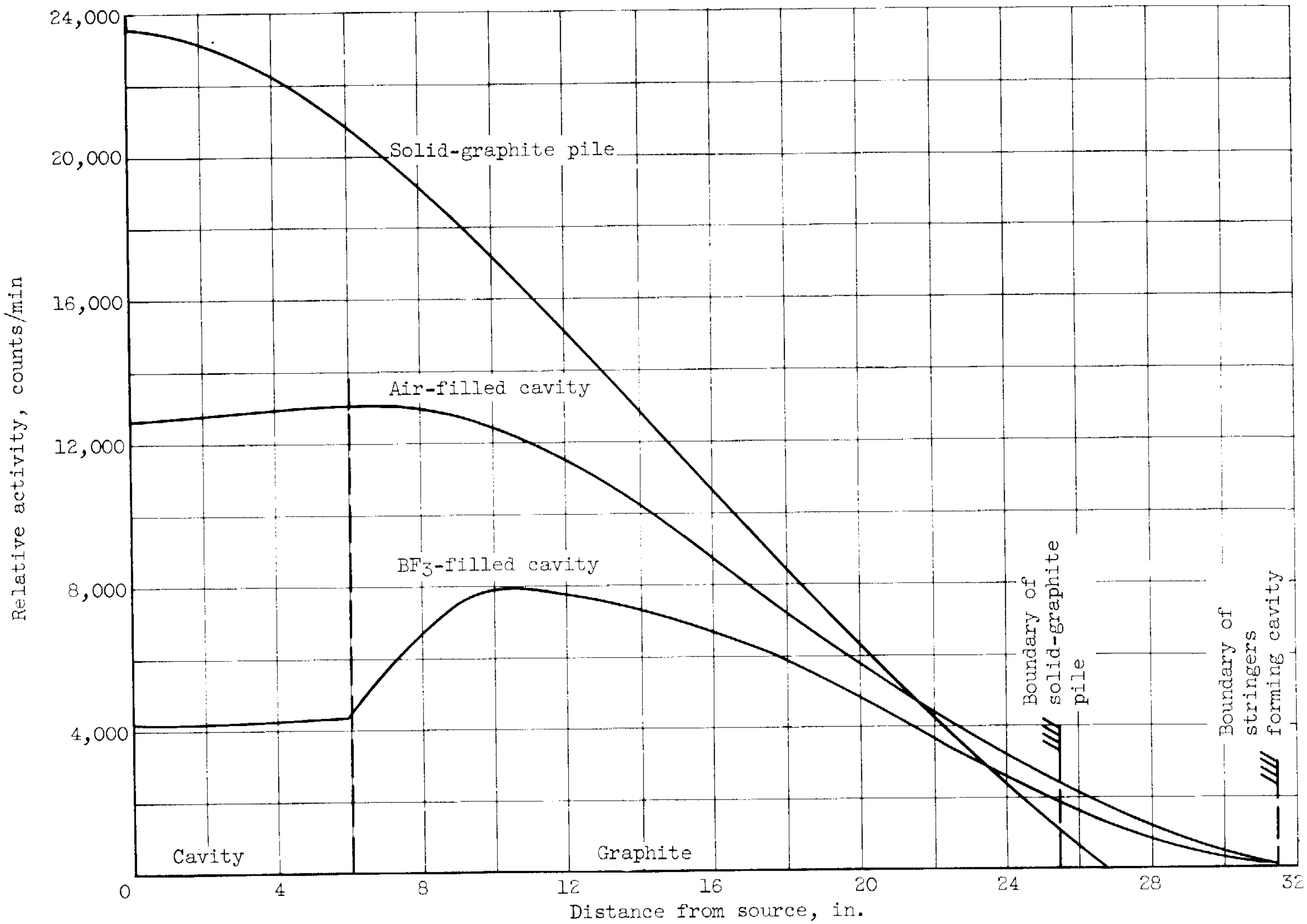

Figure 8. - Comparison of thermal flux distributions in cavity and noncavity cases.

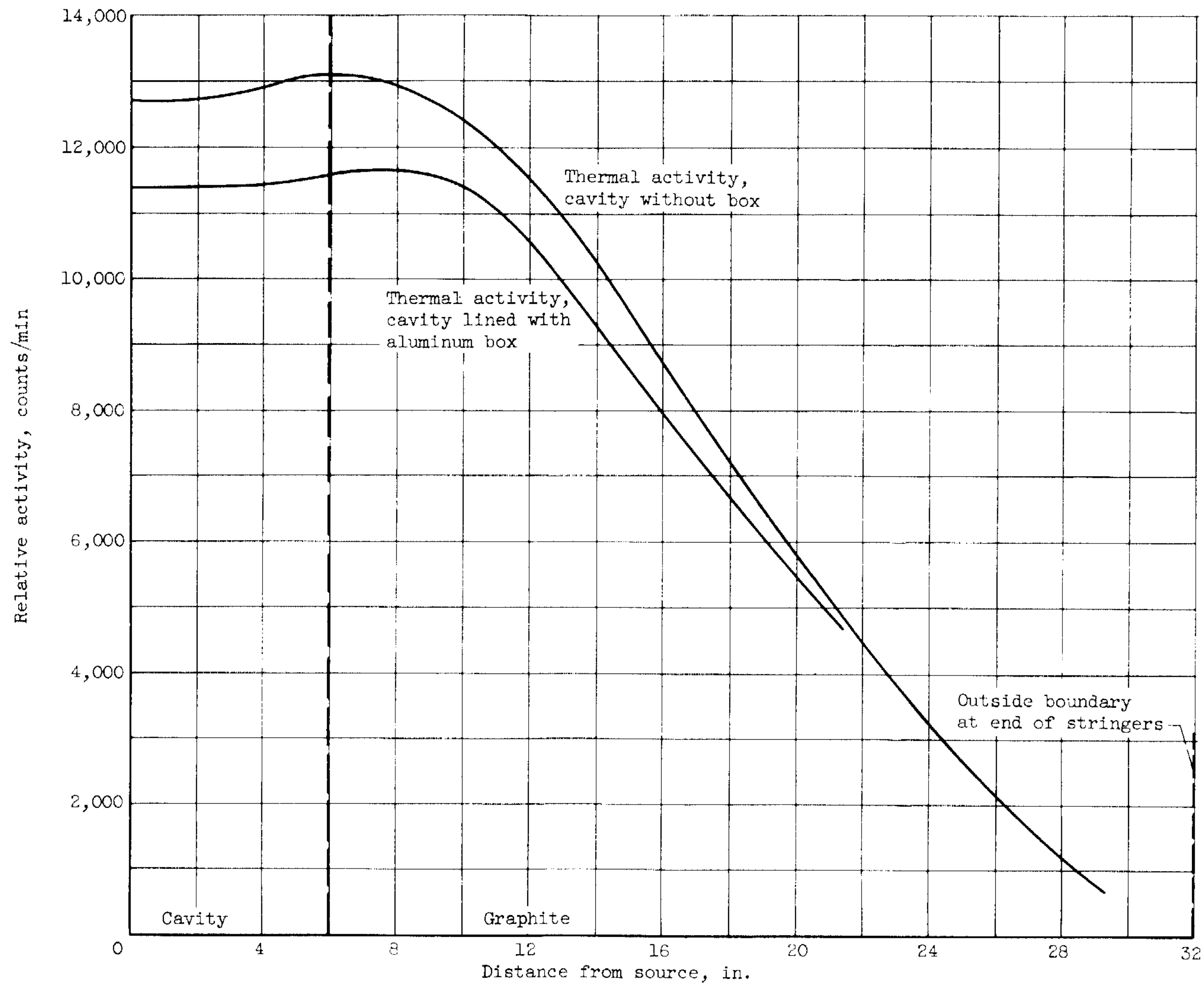

Figure 9. - Thermal neutron horizontal traverses with and without aluminum box in 12-inch cubic cavity.

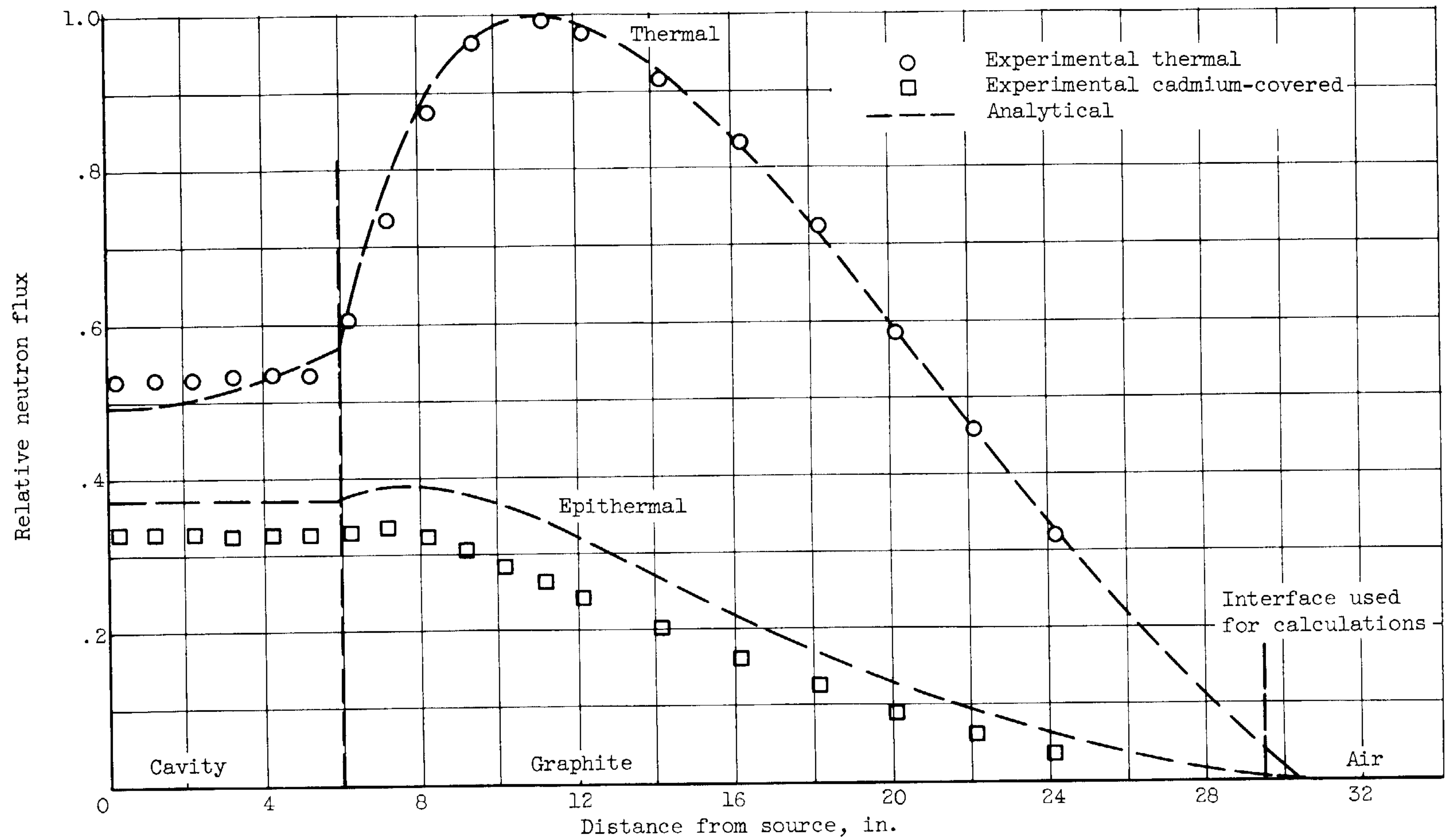

Figure 10. - Comparison of experimental and analytical flux distributions for BF_3-gas-filled cavity case.